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W3F1-2001-0061
A4.05
PR

July 23, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Request for Review and Approval of License Basis Change
Regarding GL 96-06 Over-Pressurization of Containment Penetrations

Gentlemen:

The purpose of this letter is to provide a follow-up response to a commitment identified in the NRC Staff letter dated, December 22, 2000, "Completion of Licensing Action for Generic Letter (GL) 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." Specifically, in that letter, the NRC Staff accepted Entergy Operations, Inc.'s (Entergy) commitment to resolve the ten remaining containment penetrations of an original seventeen penetrations that were potentially susceptible to thermally induced overpressurization through an evaluation, detailed analysis, or installation of physical modifications prior to startup from the Spring 2002 (March) refueling outage.

Entergy determined the permanent resolution to the GL96-06 issues for the ten remaining containment penetrations could be satisfied through the installation of thermal relief valves on three of the containment penetrations during Refuel Outage 11 and a change to Waterford 3's license basis for the seven other containment penetrations. Therefore, in accordance with 10CFR50.90, NRC Staff review and approval is requested for a deviation to the Waterford 3 licensing basis commitment to comply with ASME Section III Code, Class 2 design provisions for the containment penetration piping, as described in the Final Safety Analysis Report (FSAR). Consistent with the NRC Staff response to a previous Entergy ASME Section III

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Code change request documented in a NRC letter from C.P. Patel to C.M. Dugger, dated September 17, 1997, this change request is being submitted as a licensing basis amendment, as opposed to a 10 CFR 50.59 evaluation or a Request for Relief pursuant to 10 CFR 50.55a.

These seven penetrations are potentially susceptible to thermally induced overpressure conditions because they could contain fluids at temperatures below the 260°F Design Basis Accident maximum containment temperature during plant heatup in Mode 4. During Modes 1, 2, and 3 two of these penetrations would normally be in service and the other five penetrations would be in service intermittently to obtain process samples. These penetrations would contain process fluid at temperatures representative of the reactor coolant system and not experience thermally induced overpressurization during a DBA.

The bases for requesting this license basis change is provided through implementation of administrative procedural controls, risk analysis, and engineering analysis. The administrative procedural controls implemented ensure:

(1) The subject penetrations/systems are in service and flowing or flushed with process fluid at temperatures representative of reactor coolant during plant heatup prior to entering Mode 4 (200°F) and Mode 3 (350°F) and

(2) The containment isolation valves (CIVs) for the five sample line penetrations will be closed to minimize process fluid cooldown, when the process fluid samples have been obtained during normal plant operation and the laboratory sample valves downstream of the CIVs are closed or flow through the penetration has stopped.

The risk analysis indicates the plant risk impact involving the probability of a Design Basis Accident (DBA) and simultaneous containment penetration failures in Mode 4 met the risk acceptance guidelines contained in Regulatory Guide 1.174. Engineering analysis performed on these penetrations determined allowable stresses did not exceed the ASME Section III Code, Appendix F pipe stress values, therefore, the penetrations remain operable. This engineering analysis supporting operability remains valid.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and it has been determined that this request

involves no significant hazard considerations.

Entergy requests the NRC Staff approval of a revised Waterford 3 resolution of the remaining seven containment penetrations potentially susceptible to thermally induced overpressurization prior to startup from the Spring 2002 (March) refueling outage. Although this change request is neither exigent nor an emergency, your prompt review is requested.

All the commitments contained in this submittal are identified on the attached commitment identification form. These three commitments will replace the remaining Entergy commitment identified in the NRC Staff letter dated, December 22, 2000, to resolve the ten remaining containment penetrations prior to startup from the Spring 2002 (March) refueling outage. Should you have any questions or comments concerning this change request, please contact Ron Williams at (504) 739-6255.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 23, 2001.

Very truly yours,

for

J.T Herron
Vice President, Operations
Waterford 3

JTH/RLW/cbh

Attachments:
1. License Basis Change Regarding GL 96-06 Over-Pressurization of Containment Penetrations
2. Marked-up FSAR Pages
3. Commitment Identification/Voluntary Enhancement Form

cc: E.W. Merschoff, NRC Region IV, N. Kalyanam, NRC-NRR, J. Smith, N.S. Reynolds, NRC Resident Inspectors Office, Louisiana DEQ/Surveillance Division, and American Nuclear Insurers

ATTACHMENT 1
To W3F1-2001-0061

License Basis Change Regarding GL 96-06
Over-Pressurization of Containment Penetrations

License Basis Change Regarding GL 96-06
Over-Pressurization of Containment Penetrations

In accordance with 10CFR50.90, NRC Staff review and approval is requested for changes to the Waterford 3 license basis for the containment penetration piping, as described in the Final Safety Analysis Report (FSAR). The specific containment penetrations addressed in this change request are the portions of Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System that penetrate the containment. The FSAR specifies that the containment penetration piping and isolation valve design and fabrication meet the intent of ASME Code, Section III, Class 2 provisions (FSAR 3.2.2). The effective ASME Code for Waterford 3 is the 1971 Edition with Addenda through Winter 1972. Within Section III, Class 2 code provisions, subsection NC-3621.2 specifies piping systems that have the potential to experience pressurization due to trapped fluid expansion shall be designed to withstand the increased pressure or have provisions for relieving the excess pressure. Specifically, Entergy proposes to revise the FSAR license basis for Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System containment penetrations to identify a Waterford 3 deviation to ASME Section III, Class 2 code provisions. This deviation is required for the above specified system containment penetrations because of the potential to exceed ASME Section III, Subsection NC 3600 code allowable piping yield stresses.

DESCRIPTION OF PROPOSED CHANGES

The following proposed changes would be added to the Waterford 3 FSAR Table 3.2-1, Classification of Structures, Systems, and Components. These sections specify the license basis for the seven containment penetrations being addressed in this submittal, specifically 4-inch diameter Steam Generator Blowdown System penetrations #5 and #6; ½-inch diameter Primary Sampling System penetrations #28, #29 and #30; and ½-inch diameter Secondary Sampling System penetrations #52 and #68.

FSAR Section	Description
Table 3.2-1 Sheet 23 of 26 Containment Piping Penetrations	Insert number "29" to the Containment Piping Penetrations row under the "Notes" column
Table 3.2-1 Sheet 26 of 26 Containment Piping Penetrations	Note 29 - Thermal relief valves are not installed on containment penetrations 5, 6, 28, 29, 30, 52, and 68 for protection against potential thermally induced overpressurization during a DBA as addressed in GL 96-06.

BACKGROUND

Description of Structure, System, or Component

The Containment Isolation System (CIS) provides a means for isolating fluid systems that pass through the containment in order to confine any radioactivity that may be released in the event of a Design Basis Accident (DBA) inside the containment building. The steel containment vessel and all the penetrations are designed to withstand the pressure and temperature transients calculated to exist after a DBA. The DBAs include LOCAs, Main Steam Line Breaks, and Feedwater Line Breaks.

The piping penetration assemblies are seismic Category 1, Safety Class 2. They provide for the passage of process, service sampling, and/or instrumentation pipelines into the Reactor Containment Vessel while maintaining containment integrity. They also provide for adequate movement between pipes and the containment structure during all phases of operation and accident conditions.

In order to maintain containment integrity, all containment penetrations have the following design characteristics:

- In the unlikely event of the postulated rupture of the RCS or secondary piping inside containment, the penetrations will withstand the maximum internal pressure that would occur from such an accident.
- Penetrations are capable of withstanding the jet forces associated with the flow from a postulated rupture of a pipe in the penetration adjacent to it, while still maintaining containment integrity.
- Penetrations are capable of safely accommodating all thermal and mechanical stresses that may be encountered during all modes of operation.

The description of the piping system for the seven containment penetrations addressed in this license basis change request is as follows:

1. Steam Generator Blowdown System

   The Steam Generator Blowdown System is used in conjunction with the Sampling System to control the chemistry of the water on the secondary side of the steam generators. By maintaining high purity water in the steam generators, the buildup of corrosion products and fouling of the steam generator heat transfer surfaces is reduced.

A blowdown line from each Steam Generator removes secondary water for processing to maintain steam generator chemistry requirements. The two-blowdown lines exit the containment through separate penetrations (i.e. Penetrations #5 & #6). Pneumatically operated containment isolation valves are provided on each side of the penetration for closure on a Containment Isolation Actuation Signal (CIAS) ensuring containment integrity.

Air operated flow control valves located downstream of the containment isolation valves reduce the fluid pressure and temperature leaving the steam generators from 920 psig and 506°F to maximum values of 50 psig and 280°F. This fluid then enters the blowdown flash tank, where it flashes to steam. The water is discharged to the blowdown heat exchangers and blowdown demineralizers, then returned to the steam cycle via the main condenser.

2. Primary Sampling System

The main purpose of the Primary Sampling System is the collection and analysis of fluid and gaseous grab samples from the Reactor Coolant System (RCS), Chemical and Volume Control System (CVCS), Safety Injection System (SIS) and the Primary Water Storage Tank (PWST). Samples are taken during all modes of operation without requiring access to the Containment. The three RCS sample points which penetrate containment are: (1) RCS hot leg, penetration #28; (2) Pressurizer Surge Line, penetration #29; and (3) Pressurizer Steam Space, penetration #30. Each penetration has two automatic isolation valves arranged in series. Each containment isolation valve closes on a Containment Isolation Actuation Signal (CIAS) ensuring containment integrity.

3. Secondary Sampling System

The Secondary Sampling System takes water and steam samples from the secondary cycle, make-up demineralizer, blowdown demineralizer, condensate transfer pump discharge, and steam generators and brings them to a common location in the secondary lab and sampling room for analysis. The two steam generator sample lines penetrate containment through penetration #52 for SG 1 and penetration #68 for SG 2. The steam generators are automatically monitored on a continuous basis. The samples are analyzed for radioactivity, conductivity, pH, silica, and sodium. Each penetration has two automatic containment isolation valves arranged in series that close on a Containment Isolation Actuation Signal (CIAS) ensuring containment integrity.

BASIS FOR PROPOSED CHANGE

Deterministic Assessment

Generic Letter 96-06 was issued on September 30, 1996 to notify licensees about safety-significant issues that could affect containment integrity and equipment operability during accident conditions. GL 96-06 requested the licensees to determine if piping systems that penetrate the containment were susceptible to thermal expansion of fluid trapped between the inside and outside containment isolation valves caused by an increase in containment temperature during a Design Basis Accident. This thermal expansion could lead to a significant increase in the internal pressure of the piping and result in exceeding the design basis ASME Section III, Subsection NC 3600 code allowable stresses and possible failure of the penetration.

In response to Generic Letter 96-06, an evaluation of all containment penetrations was conducted. Seventeen penetrations were found to be susceptible to thermally induced pressurization. Condition Reports 97-0174 and 99-0982 were initiated to address the subject penetrations and determine equipment operability along with the identification of corrective actions. Entergy documented in letters W3F1-97-0232, dated October 17, 1997 and W3F1-99-0149, dated October 28, 1999, that seventeen containment penetrations were identified to be potentially susceptible. These penetrations could potentially jeopardize the ability of the containment Isolation valves to perform their safety function and lead to a breach of containment integrity. These penetrations were assessed for operability in accordance with GL 91-18 and concluded operable using the allowable stresses in ASME Section III Code, Appendix F for plastic deformation.

In accordance with the commitments delineated in Entergy letter W3F1-99-0149, dated October 28, 1999, Waterford 3 took the following corrective action during the Oct.-Nov. 2000 Refuel Outage 10 to resolve issues with containment penetrations potentially susceptible to thermally induced overpressurization:

1. A thermal relief valve was installed on three containment penetrations (Component Cooling Water Return from Containment pen. #24, Chemical Volume Control Letdown pen. #26, and Reactor Coolant Pump Controlled Bleedoff pen. #44) inside the primary containment.

2. System configuration changes (valve lineups) were implemented in four containment penetrations (Primary Makeup pen. #7; Fuel Pool Ion Exchanger Return pen. #51, Reactor Cavity Drain Pump Discharge pen. #62, and Demin. Water to Shutdown Cooling Vac. Priming Pump pen. #71) to ensure the affected piping either remains in a drained condition during normal plant operations or

has a relief path for thermal expansion of trapped fluids. The new configuration alleviates the potential overpressure issue for these penetrations.

3. Interim administrative procedural controls were established to ensure five (5) containment penetrations (Steam Generator 2 Blowdown pen. #5, Steam Generator 1 Blowdown Penetration pen. #6, Reactor Coolant Hot Leg Sample pen. #28, Pressurizer Surge Line Sample pen. #29, and Pressurizer Steam Space Sample pen. #30) were flushed with fluid greater than the 260°F Design Basis Accident (LOCA) maximum containment temperature listed in the FSAR. These administrative controls were implemented pending a final resolution to be developed prior to startup from refueling outage 11.

Entergy committed to develop a permanent resolution of the GL 96-06 issues for the remaining ten penetrations prior to startup from the Spring 2002 (March) refueling outage. The resolution will fall into the following two categories:

- Installation of a thermal relief valve on containment penetrations (1) Containment Sump Pump Discharge pen. #42, (2) Reactor Drain Tank Pump Suction pen. #43, and (3) Safety Injection Tanks Drain to RWSP pen. #59, during Refuel Outage 11.
- Request for a plant-specific change to the Waterford 3 license basis requirements for containment penetrations #5, #6, #28, #29, #30, #52, and #68 as described in the FSAR.

Administrative Controls

These seven penetrations are potentially susceptible to thermally induced overpressure conditions because they could contain fluids trapped within these penetrations at temperatures below the bounding 260°F Design Basis Accident maximum containment temperature during plant heatup in Mode 4. This condition could result in fluid heat up and an internal pressure increase. Note, Waterford 3 does not have a Mode 4 calculation of the maximum containment temperatures during a DBA, therefore, the resultant containment temperature due to a Large Break LOCA with the plant in Mode 1 was used to bound Mode 4 conditions.

During Modes 1, 2, and 3, the Steam Generator Blowdown lines would normally be in service and contain process fluid at temperatures representative of the reactor coolant system. If secured, Operations procedures require the Steam Generator Blowdown line containment isolation valves (CIVs) to be closed, thus trapping fluid with a temperature greater than the maximum containment temperature experienced during a DBA. The other five primary and secondary sample lines would be in

service intermittently to obtain process samples for compliance verification with Technical Specifications (TS) and Technical Requirements Manual requirements. When the process fluid samples have been obtained and the laboratory sample valves downstream of the CIVs are closed or flow through the penetration has stopped, the CIVs will be closed to minimize process fluid cooldown. If a DBA occurred during the period the CIVs are open and a containment isolation signal was initiated where both isolation valves closed, fluid temperature within the penetration would closely reflect the pre-accident temperature source, the RCS. Thus, with the trapped fluid at a higher temperature than the post accident ambient temperature, no heat up or over-pressurization of the penetrations would occur.

Therefore, since the window of vulnerability (plant heat-up Mode 4) for pressurization from thermal expansion is relatively small and the safety significance of these system penetrations is small, as previously discussed in Entergy letter W3F1-97-0149, dated October 28, 1999, administrative procedural controls have been identified as resolutions for the seven penetrations. Note, the Mode 4 window of vulnerability is limited to plant heatup only, however, for conservatism a value of 94.3 hours (1% per year) composed of both heat-up and cooldown hours was used to bound the Mode 4 duration. The administrative procedural controls will ensure:

(1) The subject penetrations/systems are in service and flowing or flushed with process fluid at temperatures representative of reactor coolant during plant heatup prior to entering Mode 4 (200°F) and Mode 3 (350°F) and

(2) The containment isolation valves (CIVs) for the five sample line penetrations will be closed to minimize process fluid cooldown, when the process fluid samples have been obtained during normal plant operation and the laboratory sample valves downstream of the CIVs are closed or flow through the penetration has stopped.

These administrative controls will minimize the heat-up of any trapped fluid in these seven containment penetrations and the potential thermal expansion that could lead to over-pressurization should a DBA occur that results in the maximum 260°F containment temperature during Modes 1, 2, 3, and 4.

ASME Section III Code, Appendix F

The seven containment penetrations were analyzed and determined that internal pipe pressure could cause ASME Section III, Subsection NC 3600 code allowable yield stresses to be exceeded. However, when evaluated for operability in accordance with GL 91-18, the containment penetration piping analysis for post yield stresses was found to be below the Service Level D stress intensity allowables

contained in ASME Section III Code Appendix F 1995, paragraph F-1341.2. In addition to meeting the Appendix F stress intensity, the internal pressure resulting from the final water specific volume at 260°F was shown not to exceed the lowest value of the calculated piping burst pressure, and the final hoop strain was below 2.6%. The results of these analyses show that the penetrations may exceed normal yield stresses and experience plastic deformation, but would not catastrophically fail (burst); therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity.

The Waterford 3 plant-specific risk impact from thermally induced overpressurization failure of the seven piping penetrations during a bounding Mode 1 DBA while the plant is in Mode 4 is evaluated in the following Probabilistic Safety Assessment section.

Probabilistic Safety Assessment

Entergy performed a risk assessment, using Probabilistic Safety Assessment (PSA) techniques, to determine the conservative contribution of thermally induced overpressurization failure mechanism of the seven piping penetrations on the Large Early Release Frequency (LERF) and Core Damage Frequency (CDF). The CDF is unchanged by the overpressurization failure of the identified piping. The change in LERF resulting from an assumed installation of relief valves on these seven containment penetrations to counteract the postulated thermal expansion overpressurization was calculated to be on the order of $1x10^{-9}$. The Waterford 3 LERF is on the order of $1.8x10^{-6}$ per year. The ΔLERF value remains well below the very small change of $1x10^{-7}$ ΔLERF given in Regulatory Guide 1.174 for a LERF of less than $1x10^{-5}$. Thus, the risk benefit due to adding relief valves is extremely small and does not warrant that plant change.

Conclusion

The proposed revisions to the FSAR license basis requirements are required for the Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System containment penetrations because of the potential to exceed normal code allowable piping yield stresses during a small window of pressurization susceptibility, specifically during plant heat up in Mode 4. The proposed change would be an acceptable permanent resolution to the GL 96-06 issue for these seven penetrations based on the implementation of the administrative procedural controls, the results of the engineering analysis performed using acceptance criteria contained in to ASME Section III Code, Appendix F 1995, and the negligible reduction in LERF

that would be achieved by adding thermal relief valve overpressure protection. The change in LERF is not risk significant and is too small to justify the addition of the relief valves.

The proposed change maintains consistency with the defense-in-depth philosophy through (1) a reasonable balance between prevention of core damage, prevention of containment failure and consequence mitigation, and (2) no over-reliance on programmatic activities to compensate for weaknesses in the plant design.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. (Entergy) proposes to revise the Waterford 3 FSAR license basis for the following containment penetrations: (1) Steam Generator Blowdown System penetrations #5 and #6; (2) Primary Sampling System penetrations #28, #29 and #30; and (3) Secondary Sampling System penetrations #52 and #68; to identify the Waterford 3 deviation to ASME Section III, Class 2 code provisions. This deviation is required for the above specified system containment penetrations because of the potential to exceed normal code allowable piping yield stresses during a small window of pressurization susceptibility during plant heat up in Mode 4.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this change request follows:

1. **Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed FSAR change reflects the use of administrative procedural controls to ensure these seven containment penetrations (two 4-inch diameter Steam Generator Blowdown penetrations and five - ½-inch diameter Process Sampling penetrations) contain fluid at temperatures representative of Reactor Coolant, and the very low probability for overpressurization failure of containment penetrations during Mode 4 plant operation as a permanent solution to the GL 96-06 issue. The engineering analysis determined these seven containment penetrations met the acceptance criteria for allowed stresses contained in ASME Section III Code, Appendix F 1995. The result of the risk analysis is such that the very small change in LERF, on the order of $1x10^{-9}$ per reactor year, remained well below the $1x10^{-7}$ ΔLERF guideline for a small change given in Regulatory Guide 1.174. The negligible reduction in LERF that would be achieved by adding

thermal relief valve overpressure protection is not risk significant and is too small to justify the addition of the relief valves.

With respect to the probability or the consequences of an accident previously evaluated in the FSAR, the proposed deviation to the existing ASME Section III code, Class 2 design provisions and operating requirements for the seven containment penetrations would not significantly increase the probability of an accident since the administrative procedural controls are being provided to: (1) minimize penetration heat-up and over-pressurization during a small window of vulnerability, approximately 1% per year of Mode 4 plant operation; and (2) minimize process fluid cooldown during normal plant operation by closing the containment isolation valves for the five sample penetrations when process fluid samples are obtained and the laboratory sample valves downstream of the CIV are closed or flow through the penetration is stopped. Also the results of engineering analyses showed that the containment penetrations may exceed ASME Section III, Subsection NC 3500 code required yield stresses and experience plastic deformation, but would not catastrophically fail; therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity.

On this basis, the proposed changes are not considered to constitute a significant increase in the probability or consequences of an accident due to:

- Administrative controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability
- The seven containment penetrations retaining their ability to perform their safety function and maintaining containment integrity in accordance with engineering analyses performed that met acceptance criteria for allowed stresses contained in ASME Section III Code, Appendix F 1995, and
- The low risk significance of overpressurization failure of the seven containment penetrations during a DBA while the plant is in Mode 4.

The proposed changes will not significantly affect the results of any accident previously evaluated. The accident mitigation features of the plant are not significantly affected by these proposed changes. The proposed changes do not add or modify any existing equipment.

Therefore, this change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change proposes a deviation to the existing ASME, Section III, Class 2 license basis requirements for portions of the Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System that penetrate the containment as a permanent solution to the GL 96-06 issues. This change involves recognition of the acceptability of administrative procedural controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability, approximately 1% per year for Mode 4 plant operation. Added assurance is provided through the engineering analysis performed on these penetrations that determined allowable stresses did not exceed the ASME Section III Code, Appendix F 1995 pipe stress values. Therefore, the change would not contribute to the possibility of, or be the initiator for any new or different kind of accident.

   The proposed change does not alter the configuration of the plant. There has been no physical change to plant systems, structures, or components.

   Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?**

   The proposed change does not involve a significant reduction in margin of safety. The existing licensing basis for Waterford 3, with respect to the ASME Section III, Subsection NC-3621.2 provisions for portions of the Steam Generator Blowdown System, Primary Sampling System, and Secondary Sampling System that penetrate the containment, is to ensure piping that has the potential to experience pressurization due to trapped fluid expansion shall be designed to withstand the increased pressure or have provisions for relieving the excess pressure piping. With the acceptance of this proposed deviation to the license basis, it will be recognized that the seven containment penetrations have administrative procedural controls to minimize penetration heat-up and over-pressurization during the small window of vulnerability, approximately 1% per year for Mode 4 plant operation. Added assurance is also provided through the engineering analysis performed on these penetrations that determined stresses did not exceed the ASME Section III Code, Appendix F 1995 pipe stress values and predicted the penetration piping would experience plastic deformation, but would not catastrophically fail. Therefore, the penetrations would retain their ability to perform their safety function and maintain containment integrity. This

deviation to license basis requirements for these seven containment penetrations is not considered to constitute a significant decrease in the margin of safety.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above and the previous discussion of the proposed change, Entergy has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

Pursuant to 10CFR51.22(b), an evaluation of the proposed license basis change has been performed. It has been determined the change does meet the eligibility criteria for categorical exclusion set forth in 10CFR 51.22 (c) (9) of the regulations. The basis for this determination is as follows:

1. The proposed FSAR change does not involve a significant hazards consideration as described previously in the evaluation.

2. This change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed change does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed change does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change does not modify the system or the manner in which the system is operated.

REFERENCES

1. USNRC Generic Letter 96-06, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," 9/30/96
2. USNRC Generic Letter 96-06, Supplement 1, 11/13/97
3. ASME Code, Section, III, 1971 Edition through Winter 1972 Addenda, Subsection NC/ND-3621.2
4. ASME Code, Section III, Appendix F 1995 Edition
5. Letter W3F1-97-0017, T.J. Gaudet, "Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," 1/28/97

6. Letter W3F1-97-0232, E.C. Ewing to US NRC, "Supplemental Responses to NRC Generic Letter 96-06 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," 10/17/97
7. Letter W3F1-97-0278, C.M. Dugger to US NRC, "Request for Additional Information (RAI) Regarding Response to Generic Letter 96-06 for the Waterford Steam Electric Station, Unit 3 (TAC NO. M96883)," 12/22/97
8. Letter W3F1-99-0149, E.C. Ewing to US NRC, "Supplement to Request for Additional Information Response to Generic Letter 96-06 for the Waterford Steam Electric Station, Unit 3 (TAC NO. M96883," 10/28/99
9. Letter W3F1-2000-0035, E.P. Perkins to US NRC, "Supplement to Request for Additional Information Response to Generic Letter 96-06 for the Waterford Steam Electric Station, Unit 3 (TAC NO. M96883)," 4/5/00
10. USNRC NUREG-0933, Revision 1, "A Prioritization of Generic Safety Issues," New Generic Issue 150, 6/30/95
11. EC-S01-003, Large Early Release Frequency (LERF) Impact of Mode 4 Design Basis Accident (DBA) Pipe Over-pressure-Generic Letter 96-06, 7/10/01
12. USNRC letter, Completion of Licensing Action for Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," TAC NO. M96883, 12/22/00
13. EC-M97-004, Rev. 2, Evaluation of Containment Penetration Piping in Response to GL 96-06, 6/8/01
14. Condition Report CR-WF3-1997-0174, initiated on 1/23/97
15. Condition Report CR-WF3-1999-0982, initiated on 9/20/99
16. Engineering Request ER-W3-99-0726-00-00
17. USNRC letter, "Response to Request for Relief from the ASME Code, Section III Requirements for Waterford 3," TAC NO. M95897, 9/17/97
18. Letter W3F1-96-0087, J.J. Fisicaro, "Request for Relief from 1971 ASME Boiler & Pressure Vessel Code Section III, Paragraph NC-7153," 7/1/96

ATTACHMENT 2
To W3F1-2001-0061

MARKED-UP FSAR PAGES

TABLE 3.2-1 (Sheet 23 OF 26) Revision 11 (05/01)

Miscellaneous (Cont'd)	Safety Class	Seismic Category	Tornado* Wind Criterion	Flood** Criterion	Notes
Containment Vacuum Relief System	2	I	b	b	7, 12, 18
FHB Bridge Crane	-	I	b	b	
Reactor Polar Crane	-	I	b	b	
Containment Piping Penetrations	2	I	b	b	
Radwaste Cask Handling Bridge Crane	-	I	b	b	
→ (DRN 00-1172)					
Maintenance Hatch Shielding Door	-	I	a	b	
← (DRN 00-1172)					
Auxiliary Control Panel (Hot Shutdown) (Table 7A--3)	IE	I	b	b	7, 18
Reactor Building Canal Liner	-	I	b	b	23
FHB Spent Fuel & Spent Fuel Cask Storage Pools & Pool Liners	-	I	b	b	23
RAB Condensate & Refueling Water Storage Pools and Pool Liners	-	I	b	b	23
Fuel Transfer Tube and Penetration Assembly	2	I	b	b	
→ (DRN 99-2514, 00-1172)					
UHS Sump Pumps (Cooling Tower Areas)	NNS	-----	b or c	b	17
← (DRN 99-2514, 00-1172)					
Decontamination - Area Structure	-	I	b	b	
Meterological data collection equip.	NNS	-	-	-	17
→ (DRN 00-1172)					
Containment - Personnel Lock, equipment hatch and escape lock	2	I	a or b or c	b	
← (DRN 00-1172)					
Annulus access lock	-	I	a	a	
Plant-safety-parameter display console (I.D.2)	NNS	-	b	b	17, 18
Engineered safety features actuation system	IE	I	b	b	7, 18
RCP Oil Collection System	NNS	-	b	b	
ATWS Mitigating Systems	IE or NNS	1 or -	b	b	26

29 — INSERT

* Tornado Wind Criterion (Section 3.3)

a. Structure or component is designed to withstand design wind.

→ (DRN 00-1172)

b. The system or components are housed within a structure and/or protected by a barrier which is designed to withstand the design wind and/or missile loads.

c. System or component failure is not credible because tornado missile induced failure modes are considered improbable. Ref. section 3.5.1.4.1.

← (DRN 00-1172)

**Flood Criterion (Section 3.4)

a. Structure or component is part of NPIS which is protected against flooding.

b. System or component is housed within another structure (NPIS) designed to protect against flooding.

ITEMIZED NOTES: (Cont'd)

15. Operational 10CFR50 Appendix B QA is applied for HEPA filter & charcoal absorber.

16. Safety and seismic considerations meet the requirements of NUREG-0737.

17. These items will receive applicable 10CFR50 Appendix B QA during the operational phase. Items/services will be procured whenever possible from a vendor with a 10CFR50 Appendix B QA program. When this is not possible, additional steps will be taken by LP&L QA, such as a detailed receiving inspection to assure that it performs its required functions.

18. Isolators that connect the protective systems (Class 1E) to the plant computer (non-Class 1E) are Class 1E and seismic Category I.

19. This table is an overview of the structures, components, and systems. Refer to the station's Master Equipment List, process and instrument diagrams, and control wiring diagrams for further detail.

20. The Hydrogen Recombiner has no piping and valves.

21. Suction pressure and lube oil pressure switches on charging pumps are Safety Class N2 (C-E Quality Class 2B).

22. The ICCI SYstem is environmentally and seismically qualified. Out-of-vessel components, i.e., cables and connectors are environmentally qualified in accordance with IEEE-323-1974. In addition, the HJTC System has been extensively tested and verified under conditions similar to what it may encounter during an ICC event. The CET's have also been tested and verified to function up to a temperature of 2300F. The ICCI System will be installed and operational prior to first cycle commercial operation. Although all cabling for the CET's is qualified to Class 1E criteria, some of the cabling is run in NNS cable trays, see Appendix 1.9A for details.

23. The canal, pools, and their liners are classified as structures and are therefore not assigned a safety class, but are designed and constructed as seismic Category I under the 10CFR50 Appendix B requirements. The liners and their nozzles are constructed to ASME Section VIII code requirements using ASME Section III materials.

24. Pressure switches PS-HV-5222 AS and BS function only to alarm on a loss of instrument air. This alarm function will not be lost as a result of a failure of the associated tubing. Therefore, the signal process tubing for these switches has been classified non-safety.

25. Detector design criteria is discussed in Subsection 6.4.4.2.

26. ATWS Instrumentation and components will receive NRC QA guidance for ATWS equipment, NRC Generic Letter 85-06, Quality Assurance Guidance for ATWS Equipment that is not safety related.

27. CEDM Exhaust Fan Motor cable/connector assemblies design criteria is discussed in Section 9.5.4.7.

28. Pressure relieving devices may have administratively controlled manual block valves installed in their downstream piping.

29. Thermal relief valves are not installed on containment penetrations 5, 6, 28, 29, 30, 52, and 68 for protection against potential thermally induced overpressurization during a DBA as described in GL 96-06.

INSERT

ATTACHMENT 3
To W3F1-2001-0061

COMMITMENT IDENTIFICATION/
VOLUNTARY ENHANCEMENT FORM

COMMITMENT IDENTIFICATION/VOLUNTARY ENHANCEMENT FORM

Attachment 3 to W3F1-2001-0061
Request for Review and Approval of License Basis Change
Regarding GL 96-06 Over-Pressurization of Containment Penetrations

COMMITMENT(S)	ONE-TIME ACTION	CONTINUING COMPLIANCE	SCHEDULED COMPLETION DATE (IF REQUIRED)	ASSOCIATED CR OR ER
After NRC Staff approval of this license basis change, the FSAR will be revised as indicated in Attachment 1 to support resolution of the GL 96-06 issue for seven containment penetrations.		X	Prior to startup from Refuel 11 scheduled March 2002	ER-W3-99-0726-01-00 and CR-97-0174
Prior to startup from Refuel 11 install a thermal relief valve on the following containment penetrations (1) Containment Sump Pump Discharge pen. #42, (2) Reactor Drain Tank Pump Suction pen. #43, and (3) Safety Injection Tanks Drain to RWSP pen. #59, during Refuel Outage 11 to support resolution of the GL 96-06 issue.		X	Prior to startup from Refuel 11 scheduled March 2002	ER-W3-99-0726-01-00 and CR-97-0174
To support resolution of the GL 96-06 issue for seven containment penetrations, administrative procedural controls shall be implemented to ensure (1) the subject penetrations/systems are in service and flowing or flushed with process fluid at temperatures representative of reactor coolant during plant heatup prior to entering Mode 4 (200°F) and Mode 3 (350°F) and (2) for the five sample line penetrations, when the process fluid samples have been obtained during normal plant operation and the laboratory sample valves downstream of the containment isolation valves (CIV) are closed or flow through the penetration has stopped, the CIVs will be closed to minimize process fluid cooldown.		X	Prior to startup from Refuel 11 scheduled March 2002	ER-W3-99-0726-01-00 and CR-97-0174